**Fifth International Information Exchange Forum**
**SAFETY ANALYSIS FOR NPPs of VVER and RBMK TYPES**
16-20 October 2000: Obninsk, Russian Federation

**Development of Ignalina's Safety Analysis Reports**

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## 1. INTRODUCTION

The Ignalina Nuclear Power Plant is Lithuania's only nuclear power plant. The plant consists of two units, commissioned in December 1983 and August 1987. Both units are Soviet designed RBMK-1500 reactors and are different from the RBMK-1000 ones operating in Russia and Ukraine, having a larger nominal capacity (design capacity of one unit is 4800 MW thermal) and specific design features.

Operating nuclear power plants require a safety analysis report, which confirms the original design basis and describes the behavior of the plant for all potential accidental conditions. In accordance with regulatory requirements, the safety analysis should be based on the current status of the systems, structures and components of the NPP, and should consider all the modifications carried out during upgrading outages including those changes that are committed for implementation. For the Ignalina NPP this information is presented in several reports. Since commissioning of the Ignalina NPP a number of the safety analyses have been conducted. These include the Technical Safety Justification Report (TOB) [1], the Safety Analysis Report (SAR) [2] and its review (RSR) [3], level 1 Probabilistic Safety Assessment (Barselina) [4] and Evaluation of the RBMK-1500 Accident Confinement System [5]. A number of safety analyses and safety cases have been recommended by SAR and RSR teams and have been produced.

The Russian design institute, RDIPE, performed the initial safety studies. For the evaluation of plant response for different accidents and transients Russian developed computer codes which were never been widely validated to demonstrate that its are adequately represent a reality have been used. System Description and Accident Analysis limit issues discussed in the TOB [1]. The RDIPE calculations were performed before 1988 and therefore used the design thermal power level of 4800 MW. However, after the Chernobyl accident the maximum permissible thermal power level of Ignalina reactors was reduced up to 4200 MW. Due to these limitations a number of international studies related to the different safety aspects of Ignalina NPP have been initiated after Lithuania restore its independence and Ignalina NPP come to its jurisdiction.

## 2. IN-DEPTH SAFETY ASSESSMENT OF THE IGNALINA NPP

An in-depth safety assessment of the Ignalina NPP was undertaken and as a result a Safety Analysis Report has been produced [2] and reviewed [3]. A plant-specific Safety Analysis Report is produced which will form the basis for decisions on future operation of Ignalina NPP. The SAR aims to:

- assess the safety level of the plant through an analysis and its review comparable to that commonly performed for Western nuclear power plants,
- identify and evaluate any factors which may limit the safe operation of the plant in the foreseeable future,
- assess the Ignalina NPP safety standards and practices,
- recommend any additional improvements which are reasonably practicable and provide estimates of their cost and schedule.

The safety analysis will consider a safety assessment of both units at the Ignalina NPP. The main reference plant for the project is unit 1, but a survey is included which defines the differences between unit 1 and unit 2 and assesses their safety. The assessment consists of two elements: Safety Analysis Report and an independent Review of Safety Report. The report was Ignalina NPP responsibility, supported by RBMK design institute, RDIPE and Western engineering companies. The review was undertaken by Western and Eastern technical support organizations, including Lithuanian Energy Institute.

The clear separation of the SAR production and its independent review performed in parallel and providing interactive feedback has proven very effective in ensuring an objective in-depth assessment. The SAR and RSR teams have identified safety issues and make recommendations on necessary safety improvements in design, operation and safety culture required as sound basis for plant operation.

As noted above, the SAR was initially conceived as a Western-style safety analysis report, but the completion of such a SAR would have consumed several times the resources budgeted for the in-depth safety assessment of Ignalina NPP.

## 3. FOLLOW-UP SAFETY ANALYSES

In the view of the results of the accident analysis, assessment of capabilities of the existing systems and of safety management practices produced in the SAR [2], and with expeditious implementation of all of modifications, procedures, and processes identified in the report, the SAR team supported the Ignalina NPP management convincing that:

- an adequate safety case for continued operation of Ignalina NPP had been demonstrated,
- the safety case would be adequate to the point of first gas gap between pressure tubes and graphite stack closure, which would be the life-limiting factor, and
- the plant's safety standards and practices had been assessed and recommendations for improvements had been made and accepted by the Ignalina NPP.

Recommendations for safety enhancement measures stated in the SAR [2] include not only hardware implementation at the Ignalina NPP but also further analysis to be performed. The most important recommended safety analyses are as follows:

- Safety Case for CPS/ECCS
- Safety Case for the structural integrity of the ACS
- Safety Case for the structural integrity of the Reactor Cooling Circuit (RCS), including assessment of waterhammer effect on ECCS/GDH check valves and connected pipelines
- Support analysis for the compensatory measures for CPS deficiencies
- Justification of omission of an assessment of accident at shut-down reactor
- Justification for category of accidents initiated by equipment failure omitted from analysis
- Analysis of reactivity initiated events for core with new fuel design
- Safety Cases in support for the implementation of early reactor trips and ECCS actuations (based on low flow in one GDH, low reactivity margin, and dP/dt in steam separator signals)
- Partial ATWS analyses
- Accident analysis in the long term including accidents during reactor shut-down, internal area events and external events
- Strategy for local flow degradation in intermediate and long term development

Below are presented brief summary of some analyses completed recently.

### 3.1 SAFETY CASE FOR CPS/ECCS

Ignalina NPP has been fully responsive to these recommendations and initiated the effort to perform a detailed and comprehensive Single Failure Analysis [6] and prepare a safety case. The work was performed by a team of analysts from the Lithuanian State Information Technology Institute, with significant technical input from the Instrumentation and Control Department at Ignalina NPP, and with external guidance from Swedish experts (ES-Konsult AB). The scope of the analysis produced focuses (as originally intended) on single failures arising from internal faults within the CPS-EPPS-TITAN systems and associated support systems (e.g. power supplies, ventilation). Very detailed analysis has been performed to find out whether failure of a single component could cause a loss of safety function. Due to potential for severe consequences the shutdown function is of utmost importance. External faults (such as fire and seismic) while acknowledged to be important, are being dealt with via other Ignalina safety improvement program [7] efforts currently under way and are not as extensively dealt with in the study.

The review of this study consisted of detailed review of the Single Failure Analysis documentation by a team consisting of members of the original Ignalina RSR team including experts from the Ignalina Safety Analysis Group and Western organizations. Summarizing the major conclusions and findings [8]:

- The review found that the Single Failure Analysis (SFA) was carried out in compliance with the recommendations of the RSR and Ignalina Safety Panel (ISP) and used the required IAEA safety guides and standards. The study considered 21 postulated initiating events which place a wide spectrum of demands on the proper functioning of the CPS/EPPS. The RSR reviewers looked at CPS/EPPS logic dealing with all 21 PIEs. The 21 PIEs chosen, were developed from the list used in the Barselina PSA Report [4]. The body of the analysis systematically looked for undetectable (latent) faults and documented the results via failure modes and effects analysis tables. The RSR reviewers were provided with all documentation requested, and answers to all technical questions, and were able to duplicate much of the analysts work. This provided high confidence in the integrity of the analysis.
- Original RSR concerns [3] regarding safety impact of AZ-1 reset logic and EPPS 40 second logic pulse/reset have all been fully resolved and the reviewers conclude there are no single failures or safety concerns.
- The RSR review of the SFA identified the issue of non-compliance with current standards [9] for analog signal isolation between CPS measurement channel signals and the TITAN system. This was expected from past safety reviews of RBMK-type reactors. The SFA clearly notes that the current analog signal interface circuits are designed to preclude a fault originating in the CPS from propagating back to the TITAN system. The circuit design uses only a 1kΩ resistor to isolate the CPS from faults originating in the TITAN system. This design is not in conformance with generally accepted Western nuclear safety standards [9]. The interfaces between CPS/EPPS and TITAN involve circuits of an older design which do not possess current day analog signal isolation devices. However, based upon information provided by the INPP it is clear that the impacts of such adverse interactions will be no more severe than the loss of a single CPS/EPPS channel - in the worst case. In view of this, the RSR reviewers have concluded the design meets the single failure criteria and is acceptable. The RSR reviewers, however, recommend that future modifications designed to improve the reliability of the CPS/EPPS (such as the DAZ system being implemented to address one of the RSR recommendation) address the most current industry standards for analog signal isolation.
- The RSR review of digital signal isolation based primarily on solid state optical isolators is acceptable and is in conformance with generally accepted Western nuclear safety standards.
- The physical separation between inputs and isolated outputs on the Relay Type “RES 8” is not in conformance with generally accepted Western nuclear safety standards. This lack of physical separation is not a new issue. The RSR review of digital signal isolation based on conventional relay circuits concludes their usage is marginally acceptable.
- The EPPS logic extensively uses “energize to trip” logic, whose availability is significantly less reliable that “de-energize to trip” logic typically used in Western designed NPPs. The availability of “energize to trip” logic, whose failures are not self-annunciating, is very sensitive to the thoroughness of the testing programs designed to detect latent faults. In this area, the Single Failure Analysis results are very sensitive to assumptions regarding the adequacy of the testing programs. The RSR reviewers performed a limited review of the test procedures for the most sensitive logic (e.g. loss of off-site power) and found that INPP apparently has sufficiently comprehensive programs in place in this area. The RSR reviewers were not able to completely review all areas - but from what was observed have confidence such programs exist and that these are being carried out at a frequency specified in the Technical Specification [10]. This was verified by a sample review of INPP testing records. The issue of “energize to trip” logic was thus concluded to be resolved as far as single failures are concerned. The RSR reviewers, however, recommend that future modifications designed to improve the reliability of the CPS/EPPS utilize “de-energize to trip” logic.
- The RSR reviewers thoroughly reviewed all 21 postulated initiating events evaluated in the CPS/EPPS Single Failure Analysis. This included detailed technical review of the submittal materials, issuing requests for further back-up documentation and schematics, and meeting several times with the analysts who prepared the study. Based on these reviews and the further responses provided by the INPP, the RSR reviewers concluded that the SFA submittal demonstrates that there are no single internal failures capable of defeating the overall CPS/EPPS functioning for the Postulated Initiating Events.
- An electrical interface circuit related single failure mode was identified in the course of the RSR review, which is potentially capable of defeating the proper functioning of the CPS/EPPS for two postulated initiating events. The circuits in question are a series of coincidence circuits (“TEZ K” modules) taken from un-isolated redundant trip channel local coincidence signals. They are brought together at one point for the purpose of performing cross-channel checks on the failure of AZ-3/AZ-4 (PIE 14) and local emergency protection (PIE 15). The coincidences were installed for diagnostic/alarm purposes - but a fault on the “TEZ K” module circuit board integrated circuits will fail all trip channels used. The un-isolated circuits were only found on the logic for protection against PIEs 14 and 15, and there is no indication the problem is present on logic for protection against other PIEs. The RSR reviewers thus recommend that Ignalina NPP

study this circuit further and recommend a suitable measure to eliminate the potential single failure in this area.

The review concluded that the Single Failure Analysis was a thorough, comprehensive analysis that exhaustively pursued the existence of potential single failures capable of defeating the overall functioning of the combined CPS/EPPS. The effort that was carried out by Ignalina NPP and their contractors was fully responsive to the recommendations of the RSR and Ignalina Safety Panel and has increased the level of confidence that the CPS/EPPS constitutes a strong line of defense. Such confidence could not be demonstrated without carrying out this work. While the reviewers conclude that the examination of the CPS/EPPS was comprehensive, this must not be interpreted to imply that the reviewers can state with absolute certainty that there are absolutely no other single failures present in the CPS/EPPS design. The reviewers do believe that there are no other obvious single failures that have not been considered based on the design information reviewed. During the course of the review, several single failures were identified and the Ignalina NPP is addressing the resolution of these. This outcome is not unexpected and is typical to safety investigations performed and reviewed for nuclear power plants throughout the world. The work was done under considerable time pressure and there was no time for the reviewers to validate all of the information of the plant that was used in the analysis. Of the single failures identified, only one was found to be potentially able to fail a system. However, justification was made by Ignalina NPP that an immediate solution is not necessary. This was supported by several arguments: the low probability of the relevant initiating events, the low probability of the single failure, very mild consequences of possible transient and the reasonable likelihood of compensating operator actions due to the slow development of the consequences. VATESI's conclusion is that operation of the plant for short-term time is permissible, but that a systematic approach to a physical resolution is required. Required hardware modifications have been installed at during 1998 outage.

### 3.2 SAFETY CASE FOR INTEGRITY OF THE ACCIDENT CONFINEMENT SYSTEM

The purpose of the project was to perform a detailed structural analysis of Ignalina NPP ACS. The Panel of Safety of Ignalina NPP demanded the realization of calculations of strength of ACS structure, on the basis of the SAR [2] and RSR [3] recommendations. Such analysis usually covers all design accidents and is an obligatory component of western SAR. For the performance of these requirements were formulated the following purposes:

- Documenting and verification of the project of system. Reconsideration of the descriptions of systems, which were performed in SAR
- Reconsideration of design calculations, which were performed by Sverdlovsk and calculations of Ignalina NPP ALS structural integrity, which were performed by Lithuanian Energy Institute earlier
- Planning and realization of test on ACS density of Units 1 and 2, performance of measures for reduction of the leak, especially, for the unit first
- Planning and realization of check of reinforcement bars for the most loaded constructions of ALS compartments
- Performance of the of the analysis of structural integrity on the real data of reinforcement bars and design pressure
- Documenting and check of an assessment of distribution of fission products on ACS compartments
- Performance of calculations of doses for design basis accidents

The complex analysis of safety of Ignalina NPP ACS, including analysis of experience of operation, engineering assessment, thermal-hydraulic and structural analysis is performed. The obtained results of calculations, results of the performed non-destructive testing and carried out experimental tests on an determining of the mechanical characteristics of concrete and reinforcement bars have not revealed essential lacks which because of would be impossible the further operation of ACS of Ignalina NPP unit 1. The structural integrity of ACS during maximum design basis accident by results of the nonlinear analysis will not be violated. For increase of a level of safety of ACS the recommendations are given.

### 3.3 SAFETY CASE FOR INTEGRITY OF THE REACTOR COOLANT SYSTEM

The main objective of the Reactor Cooling System Safety Case is to perform the detailed structural analysis of the MCC of Ignalina NPP according to the requirements of the Safety Panel of Ignalina NPP, expert groups of SAR and RSR. For performance of these requirements the following purposes are formulated:

- Development and documenting, determination of limits of the MCC and in-service inspection of the MCC;
- Development of the list of priority problems, which should be solved in the course of the project. Drawing up of the list of critical components of the MCC;

- Documenting and checking of the project of the system. Review the descriptions of systems developed in SAR by the Task Group 1;
- The analysis of the program and the results of the in-service inspection;
- Review of the original design stress calculations;
- Strength and integrity calculations of pressure tube;
- Performance of the structural analysis of the MCC of Ignalina NPP based on the real data and detailed consideration of the requirements of the in-service inspection;
- MCC components finite element stress analysis according to ASME requirements;
- Assessment of the waterhammer effect on ECCS/GDH checks valves and connected pipelines.

The most important conclusions on the reactor cooling system safety case of Ignalina NPP are discussed below. In the case of discrepancy to the requirements of the regulating documents are given the recommendations for their elimination.

- The results of the performed finite element analysis of Ignalina NPP reactor cooling system critical components according to the ASME III subsection NC under the conditions of normal operation, hydraulic tests and maximal calculational earthquake show, that there exist the sufficient safety margin of the strength and strength. Thus, their conformity to the categories of the design and operational criteria of ASME III requirements is shown.
- Non-fulfillment of the nonductile fracture criteria was obtained for the postulated corner crack in downcomer nozzle in the regime of pressing, carried out when temperature is 55 $^{o}$C. Relating to this, it is necessary to prepare technical conditions for the performance of pressing of MCC having the increased temperatures and to develop the technical means for the non-destructive control of the possible defects of such type.
- The maximal bending stresses during the seismic influence resulting from the maximal calculational earthquake existing in the central (side) supports of the drum-separators, are received in the result of linear calculations, exceed the allowable value 2,59 times. It means, that the axial moving of the drum-separators will result in plastic deformations in the central (side) supports. The conservative linear calculations were performed for the potential failure of the horizontal support, when the calculational stresses in the most loaded downcomers do not exceed the ultimate strength of the material. On the whole the structural integrity of the construction of supports of DS can be considered as being acceptable.
- For the conditions of the hydraulic tests of the feedwater nozzle, do not satisfy ASME III subsection NB requirements for primary membrane local stresses $P_l$. They by 29% exceed the yield strength of the material and equals to 59 % from the ultimate strength in hydraulic test temperature. These membrane local stresses are located in the middle of the wall. In the most loaded places of a nozzle appear plastic deformations and these places can be subjected to a low-cyclic fatigue. It is recommended to carry out the calculation of the DS feedwater nozzle for low-cycle fatigue.
- Because of the fact, that during the hydraulic tests the overload of a DS nozzle (feedwater nozzle) was obtained, it was recommended to reconsider an opportunity to reduce frequency of realization and/or the pressure of hydraulic tests.
- During waterhammer event on ECCS/GDH check valve, stresses in the elbows of the pipelines of lower water communication pipelines (LWCP) will cause plastic deformations. In a case of waterhammer on the check valve of GDH it is necessary to perform the control of LWCP end connected to pressure tube and the elbows of the pipelines of LWCP, and the connection of LWCP to pressure tubes.

The carried out complex analysis of the most important for safety components of the reactor cooling system of Ignalina NPP has not revealed shortcomings, which could become the reason for not allowing the further operation. The RBMK-1500 reactor cooling system of Ignalina NPP principally corresponds to ASME standards.

## 3.4 SAFETY CASE FOR ADDITIONAL REACTOR SHUTDOWN SYSTEM

A good example both of significant safety improvement in frame of implementation of SIP-2 and state-of-the-art codes applications for safety management of Ignalina NPP is development and implementation of an additional shutdown system DAZ. In accordance with this Ignalina Safety Panel recommendation VATESI has required Ignalina NPP to develop and implement a compensatory measures for Control and Protection System before Unit 1 will be allowed to restart from its 1998 outage. The Lithuanian Energy Institute performed an analysis that supports the selection of the input process parameters and setpoints values as well as developed accident analysis for the Ignalina DAZ system. It was shown that in case of transients with failure of the existing CPS but with activation of DAZ system reactor is adequately protected and any safety criteria are not violated.

Pressure behavior in the main circulation circuit in case of loss of off-site power supply with failure of the existing Control and Protection System but with activation of DAZ system is presented in Fig. 1. In this case pressure in the reactor coolant circuit is far below of limit pressure 10.4 MPa. Therefore, after implementation of DAZ system at the Ignalina plant ATWS would be moved from the beyond design basic accidents to design basic accident class.

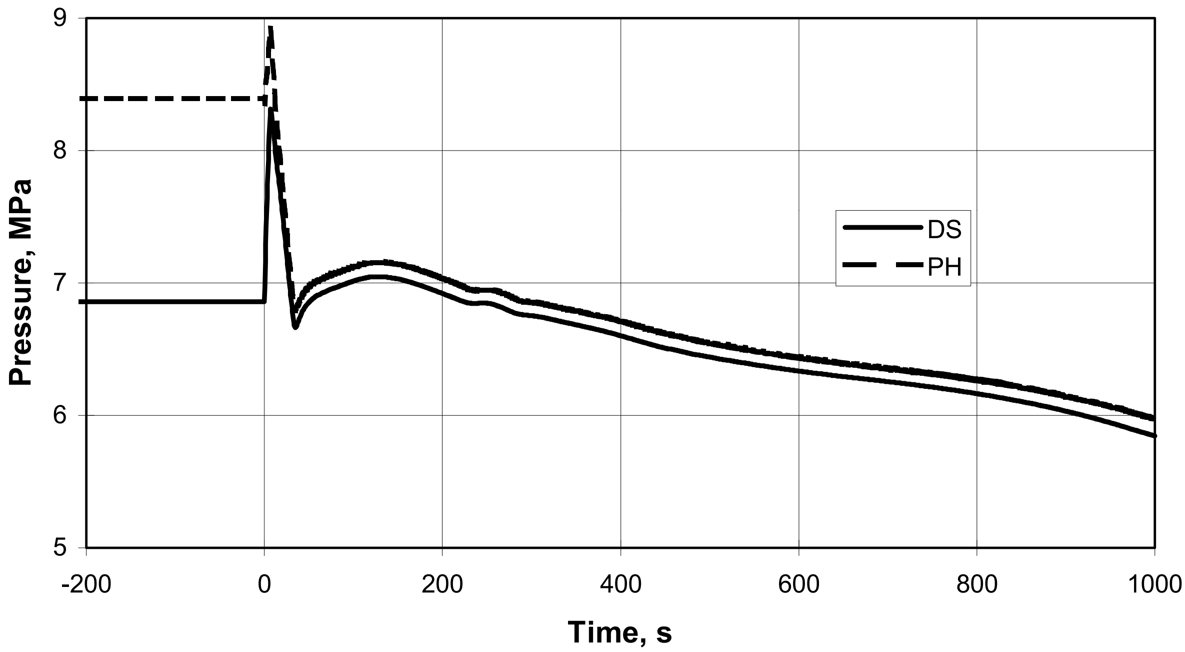


Fig. 1 Loss of off-site power supply with failure of existing Control and Protection System and activation of DAZ system. Pressure in the main circulation circuit

### 3.5 STRATEGY FOR DESTRUCTION OF FLOW STAGNATION

In any transport circuit, the coolant flows from high pressure point to a low pressure point, and it is distributed among various parallel flow paths (i.e. among the core passes, and the fuel channels within the core passes) according to hydraulic resistance of individual flow paths. In principle, breaking a downstream pipeline can transiently accelerate the flow through one or more flow paths or breaking an upstream pipeline can reverse the flow. It follows that a partial break upstream of the reactor core may also be postulated that just reduces the flow-driving pressure gradient across a flow path to near the zero. The pressure gradient may be reduced right after the break (i.e., when the decay power and the pressure are still elevated), or in longer term (i.e., when the ECCS is activated). A partial break that result in the largest power-cooling mismatch (i.e., in highest temperatures) is called the critical break.

Partial breaks in large diameter piping were not investigated at the design stage of the Ignalina NPP, simply because they were not considered credible. Once a critical flaw size is reached in the wall of large diameter pipe, it is anticipated that the break rapidly propagates to an opening that is equivalent to twice the cross-sectional area of the pipe (i.e., to the full break size). Also breaks of any single pipe connecting to the large diameter headers are not capable of producing break discharges of critical break magnitude. For this reason, the critical breaks are not realistic, but they are now included into consideration for consistency with world trends in accident analysis.

Local flow stagnation could occur in coolant transport circuit, if flow-driving pressure gradient across a flow path reduces to near zero. For RBMK type reactors the following local flow stagnation are possible:

- in one fuel channel following a partial break of one lower water communication line,

- in a group of fuel channels (i.e., between 38 and 43 channels in one core pass) following a partial break of the group distribution header or guillotine break of GDH accompanied by failure to close of the check valve at the neighboring group distribution header,
- in one loop of the coolant transport circuit (i.e., about 830 channels) following a partial break of pressure header.

The consequences of a single channel break are understood and they could lead to break of one pressure tube. This is design basis accident for RBMK type reactor, thus, a single channel break is not at issue. As to a flow deterioration in multiple channels, analyses indicate that short-term temperature excursions can indeed occur, but they tend to be relatively mild (i.e., any early flow deterioration tends to be highly transient and short-lived). Thus is because the conditions across the main circulation pumps change dramatically in the early stages of accident, so it is impossible to sustain a near zero pressure gradient across the core for any fixed break size. In the longer term, the forced convective cooling can be disrupted due to low pressure differentials between the steam separator (maintained at elevated pressure by the steaming in the intact loop) and the group distribution header (maintained at elevated pressure by the ECCS injection). In this case, no coincident failure of check valve is required to achieve a quasi-steady pressure balance. Also, because the partial break can be postulated to be of any size, different critical breaks may be required for different postulated component failures, but once the appropriate critical breaks is found for a particular plant state, the consequences are not particularly sensitive to the postulated impairments.

The smallest critical break is of interest because, in principle, the smaller the coolant discharge from the break, the longer a high system pressure can be maintained, and the longer a potential for pressure tube break exists. In the terms of the most severe plant response, a partial break in the pressure header is closely similar to that in the GDH. Since in analysis is assumed that neutronic trip signals will not be effective, so both breaks will trip on a process signal (i.e., pressure in the reinforced compartments) There is a brief power-cooling mismatch as the flow stops in the core. The pressure header breaks causes the check valves in all GDHs to close and there be no longer power cooling mismatch unless a failure of check valve is also postulated. With this failure, a single core pass is affected, i.e., the same as in case of critical break in GDH. However, the system de-pressurizes faster for pressure header break than for the GDH break, so any potential for consequential pressure tube break can only be higher for the latter break.

Based on the precluding considerations, a critical break in the GDH downstream check valve is selected as the worst partial break to be examined in this analysis. At issue of this break is the power-cooling mismatch in the early stages of accident (i.e., in the first few seconds) as well as in the long term (i.e., after the ECCS is activated).

The consequences of critical break in the GDH downstream check valve, which is selected as the worst partial break to be examined in accident analysis, are evaluated below. Stylized simulations as a partial break that impose a near-zero pressure differential across one core pass in one loop quantify the critical break size. The stylized simulations indicates that GDH breaks with openings between 17 and 20 percent of its cross-sectional area could possibly result in a period of degraded coolant flow in the broken core pass right after reactor scram. Although commonly used, the break size is not the appropriate parameters to use for the partial break classification. The discharge predictions can vary for same break flow area depending on the break geometry, discharge model parameters and other model characteristics. Therefore, the initial break discharge rate is used to represent partial breaks of different sizes. The initial discharge rate is taken to be that at one second into the accident. It was found that for fixed partial breaks, the break causing the most severe early flow degradation is the break with initial discharge rate of 1135 kg/s. Such break is taken to be the critical break of the group distribution header.

Plant response in case of any transient or accident highly depends from initial plant conditions prior to accident. The largest potential for a power-cooling mismatch is when critical GDH breaks occurs at the maximum operating power, i.e. when the reactor operates at 4200 MW prior to accident. Correspondingly to this the power level of maximum loaded fuel channel is 3.75 MW, average loaded channel is 2.53 MW and minimum loaded channel is 0.88 MW. Other initial plant process parameters are also correspond to assumed initial reactor power.

The reactor is assumed to operate at its pre-accident power level of 4200 MW until a reactor scram signal is issued by the Control and Protection System upon Emergency Process Protection System signal that the pressure in the reinforced compartment has exceed 2 kPa gauge. The fast acting scram system is activated in 1.2 seconds after critical break in the GDH is occurred. Emergency Core Cooling System is activated by EPPS signal indicated that water level in steam separators reaches “-1000 mm” mark. Main Feedwater Pumps of accidental loop are tripped when water level in deaerators drops below 0.1 m. It is assumed that Emergency Feedwater Pumps are switched off in case of pressure in their heads drop below 5 MPa.

Results of analysis of the critical break in the GDH accident shown that coolant discharge through the break during first 5 seconds of accident increase from initial value of 1135 kg/s to about 1300 kg/s, but later decrease slowly. Pressure in the accidental Group Distribution Header sharply drops from 8.3 MPa to about 7.0 MPa and already during the first seconds of accident pressure difference between this GDH and steam separator reduces to near zero. From about 300 seconds of initial break after MFPs trip pressure in accidental GDH exceeds pressure in steam separator, but after trip of the MCPs of accidental loop at time moment of 1200 seconds mentioned above pressure difference again reduces to near zero. The water levels in steam separators drop initially due to void shrinkage following reactor scram, but they recover as the feedwater continues to be injected by the MFPs. The recovery of water level in steam separators causes then the Main Feedwater Pumps trip. This leads to the fast decrease of the water level in steam separators. After 410 seconds from the initial rupture it reaches “-1000 mm” mark and Emergency Core Cooling System is actuated. ECCS water is injected both from ECCS accumulators and by ECCS pumps. Until MCPs are in operation flow rates in intact and accidental loop change insignificantly, except the channels connected to the accidental GDH where flow rates sharply drop and during first 270 seconds are fluctuated near zero Fig. 2. Later flow rates through these channels recovered when pressure balance between the accidental GDH and the steam separator is disturbed and increased after ECCS actuation. Flow stagnation at the initial stage of accident leads to the fuel cladding and pressure tube wall temperatures excursion, Figures 3 and 4.

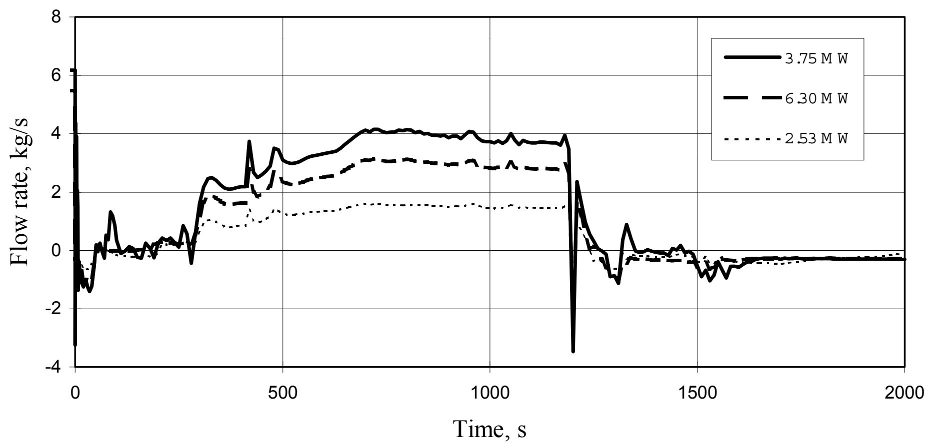


Fig. 2. Flow rate in the channels connected to the accidental GDH

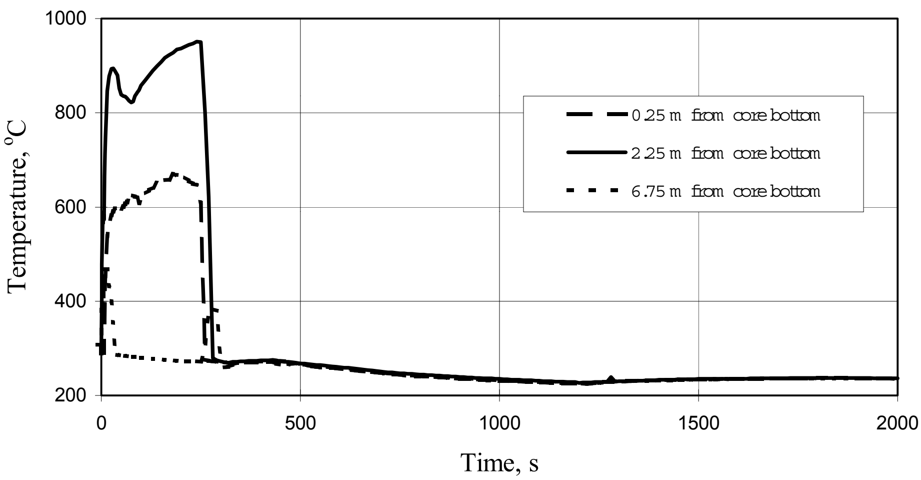


Fig. 3. Fuel cladding temperatures in the maximum loaded channel connected to the accidental GDH

Peak fuel cladding temperature in maximum loaded channel even exceeds safety criterion of 700 $^{o}$C and at the end of the first period of flow stagnation reaches 950 $^{o}$C. This could lead to cladding failures in maximum loading channel because safety criterion is exceeded during 200 seconds. Peak cladding temperatures in average and minimum loading channels also are reached in about 15 seconds from initial break, but their values does not exceed safety criterion. Peak pressure tube wall temperature in maximum loaded channel reaches 670 $^{o}$C after about 200 seconds after initial break. In spite of that safety criterion of 650 $^{o}$C is exceeded insignificantly and only during time period about 60 seconds pressure tube failure of maximum loaded channel cannot be ruled out, because this time period pressure in the coolant circuit is quite high (about 6 MPa). Peak pressure tube wall temperatures in the average loaded channel not exceed 500 $^{o}$ C and in the minimum loaded channel even decreases during the first period of flow stagnation. Second period of flow stagnation starts at about 1200-1300 seconds from the beginning of accident.

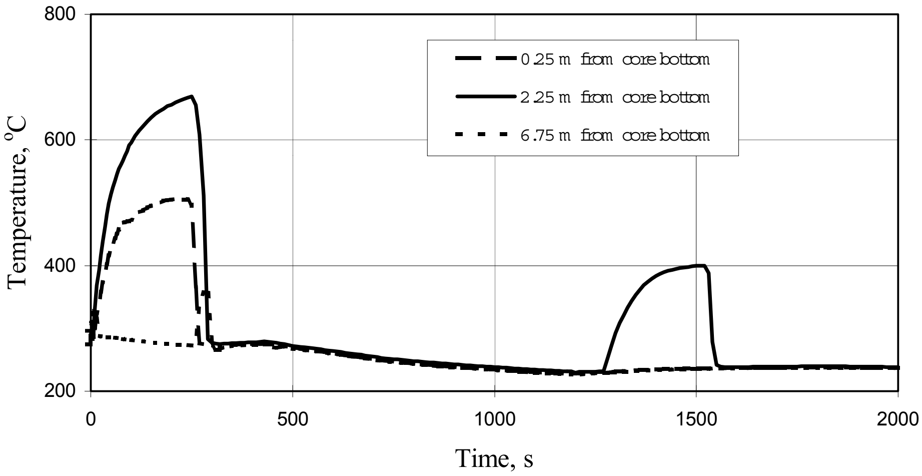


Fig. 4. Pressure tube wall temperatures in the maximum loaded channel connected to the accidental GDH

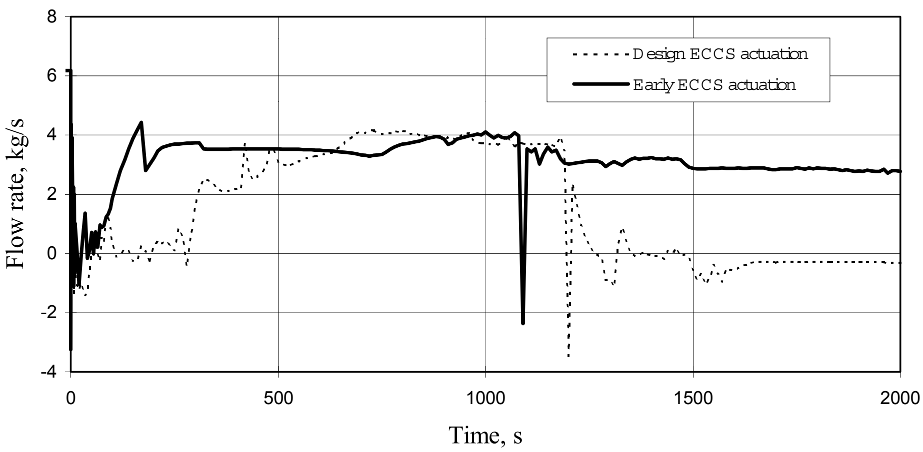


Fig. 5. Flow rate in the maximum loaded channel connected to the accidental GDH in case of early ECCS initiation

This later flow stagnation is caused by MCPs trip in the accidental coolant loop and decreasing of ECCS water injection because of emptying of ECCS accumulators. In this period only insignificant fuel cladding and pressure tube wall temperatures rise is observed.

Analysis of the critical ruptures in the GDH shown that they could lead to local flow degradation and fuel cladding and pressure tube wall temperatures excursion. So, for existing ECCS actuation logic fuel cladding and pressure tube failures cannot be ruled out during such or similar accidents. Therefore, development of strategy for destruction of local flow stagnation is required both in the early stages of accident as well as in the long term.

This includes determination of the necessary actions such as ECCS injection management or main circulation circuit de-pressurization as well as justification of the new early ECCS initiation signal, which compensates for stagnation flow. Three different possibilities for destruction of local flow stagnation have been investigated:

- early ECCS actuation,
- increasing of water injection from the ECCS pumps, and
- de-pressurization of main circulation circuit by opening of SDV-A.

According to new ECCS actuation logic which will be shortly implemented at the Ignalina NPP this system will be activated by coincidence of the signal of pressure increase in the reinforced compartment with signal on low flow in GDH. This allows early actuation of ECCS, i.e. in the first few seconds after initial break will occur. Results of analysis of the critical break in the GDH assuming that ECCS is actuated simultaneously with reactor scram system are presented in this section. In case of early actuation of ECCS rapid pressure decrease in the accidental GDH is observed. This is caused by cold water delivery by ECCS accumulators to GDHs of accidental loop. However, pressure in the steam separator also decrease faster as in the case of the design ECCS actuation. During the first 100 seconds of accident pressure difference between the accidental GDH and the steam separator reduces to near zero, but later pressure in the accidental GDH visible exceeds pressure in the steam separator. In case of early ECCS actuation MCPs of accidental loop do not trip because of sufficient water amount in steam separators. This cause that flow rate in the channels connected to the accidental GDH fluctuated near zero only during first 60 seconds of accident and later it increase up to 1.5-3.5 kg/s, Fig. 5. Peak fuel cladding temperature in the maximum

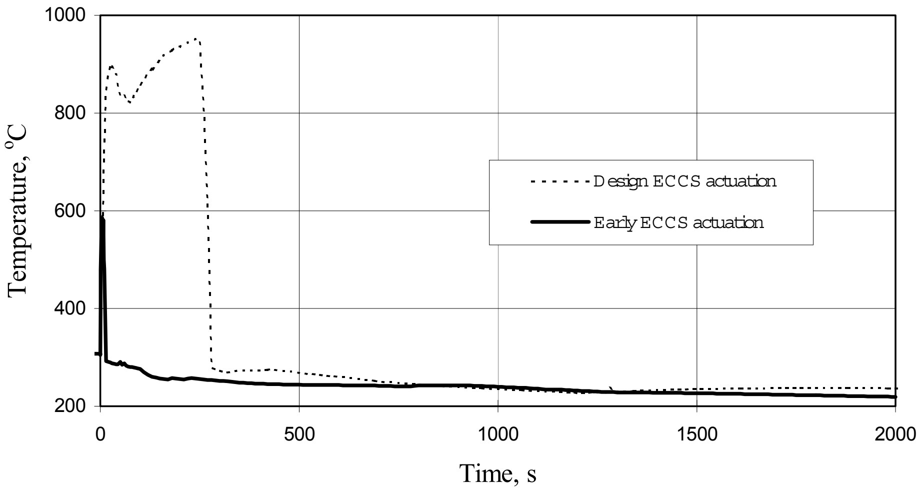


Fig. 6. Peak fuel cladding temperatures in the maximum loaded channel connected to the accidental GDH in case of early ECCS initiation

loaded channel of accidental GDH not exceeds 600 °C, Fig. 6, while fuel cladding temperatures in the average and minimum loaded channels continuously decrease. Pressure tube wall temperatures in all fuel channels connected to the accidental GDH slowly decrease, Fig. 7. Thus, early ECCS actuation compensates for stagnation flow in case of critical break of GDH both at short and long term of accident and fuel cladding and pressure tube wall temperatures never exceed safety criteria.

Flow stagnation in case of the guillotine break of GDH with failure of check valve at neighboring GDH observed at the later stage of accident after the water margin in the ECCS accumulators is exhausted. In this case increasing of water injection from the ECCS pumps as a compensatory measure for destruction of local flow stagnation should be used. De-pressurization of main circulation circuit by opening of SDV-A in this case is a little effective. Using this later measure flow stagnation would be disturbed not earlier as after 5 minutes from opening of SDV-A by operator.

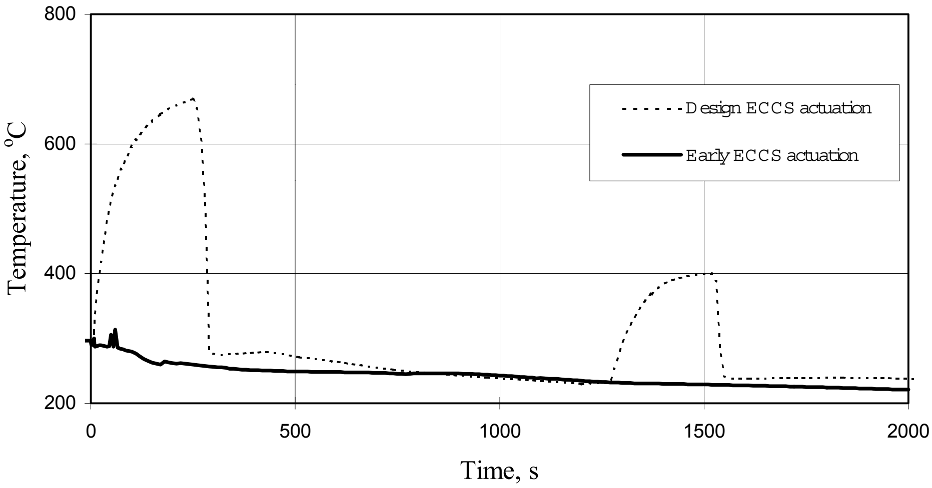


Fig. 7. Peak pressure tube wall temperatures in the maximum loaded channel connected to the accidental GDH in case of early ECCS initiation

## 3.6 SAFETY ANALYSIS OF IGNALINA NPP DURING SHUTDOWN CONDITIONS

The accident analysis for the Ignalina NPP with RBMK-1500 reactors at normal operating conditions and at minimum controlled power level (during startup of the reactor) has been performed in the frame of the project "In-Depth Safety Assessment of the Ignalina NPP", which was completed in 1996. However, the plant conditions during the reactor shutdown and at shutdown conditions differ from conditions during reactor operation at full power (equipment status in protection systems, set points for actuation of safety and protection systems, etc.). Therefore, safety analysis review team has recommended to perform analysis of accidents during reactor shutdown and at shutdown conditions. The main goal of these analyses are to demonstrate that Ignalina NPP status during low power operation and reactor shutdown bounded by plant status for similar transients during full power operation and/or lead to safe reactor conditions.

Results of RELAP5 simulation of two worst initiating events during reactor shutdown - Pressure Header rupture in case of steam reactor cooldown as well as Pressure Header rupture in case of water reactor cooldown are discussed below.

Full breaks produce maximum coolant discharges from their respective pipelines, and thus the most stringent requirement for coolant makeup. The fastest coolant loss from the heat transport system is achieved by postulating a guillotine rupture of largest diameter pipe. A guillotine rupture of the pressure header is chosen as the worst full break in the Ignalina NPP circulation circuit. Furthermore, if a coincident failure of a check valve is postulated in one of the group distribution header, a group of channels connected to this header may not be adequately cooled.

This section evaluates the consequences of guillotine ruptures of Pressure Header with failure of GDH check valve to close during reactor shutdown in cases of steam as well as water reactor cooldown whose are selected as the most severe events to be examined in accident analysis.

According to the technological regulation [10], before reactor shutdown, it is necessary smoothly to reduce the reactor power and to unload turbines. With achievement of a level of the reactor power equal 1000 $MW_{(th)}$ (the pressure in DS is equal 6.86 MPa), the reactor shut down starts by the pushing AZ-1 button. Cooling down of the reactor and MCC after turbines switching-off can be started by smooth reduction of pressure in DS with nominal levels of water at the expense of regulating steam discharge through SDV-C and SDV-D. Such cooling down of the reactor refers to as steam cooldown and continue before decrease of water temperature in the MCC up to 180 $^oC$ (pressure in DS is equal 1 MPa) [10].

It was assumed in the modelling, that after achievement of a level of the reactor power equal 1000 $MW_{(th)}$, operator shut down the reactor by pushing AZ-1 button. Simultaneously operator switches off all operating MFWP because of decrease of feedwater consumption. Because the reactor decay heat has the greatest influence on accident consequences, conservatively was accepted that the rupture occurs at the moment of a beginning of reactor power decrease. Decay heat in the reactor begins to decrease because of the CPS rods insertion into the zone and 20 seconds later after pushing AZ-1 button makes only approximately 6 % from initial power. The guillotine rupture of PH was accepted in the modelling, thus, the coolant discharges from the MCC without any interference from the MCP side and from the reactor core side through failed to close check valve on one of GDH both. Due pressure difference decrease between PH and DS in the affected loop and increase of pressure in MCC compartments, the signal on ECCS activation is generated almost at once after rupture occurs. The supplying of water from ECCS accumulators into the GDH of the MCC affected loop and from three ready to operate EFWP and four ECCS pumps into GDH of both loops of the MCC begins. The power supply for the three operating MCP in the affected loop of the MCC is disconnected by protection because of the coolant flow rate decrease on hydrostatic bearings in early beginning of the accident. MCPs of the intact loop are switched off by protection approximately in 50 seconds after rupture occurs.

The check valves of the MCC affected loop are closed after PH rupture at once. Fuel channels of this loop are cooled by ECCS water in further. The operating MCP supplied coolant in the beginning of the accident cools the channels of the intact loop of the MCC. After switching-off MCP the GDH check valves in the intact loop of the MCC are closed and FC are cooled only by ECCS water. As it is visible from Figure 8, in fuel channels connected to GDH with fail to close check valve, the coolant flow reverse appears (the coolant gets from DS and discharged through the rupture). In the beginning of the accident these FC are cooled by steam-water mixture. However, after approximately 50 seconds after the beginning of accident, the DS gets empty and these FCs are cooled by saturated steam only. By change of cooling conditions is possible to explain behaviour of fuel cladding temperatures (Figure 8). During the first minute after beginning of the accident the temperatures decreases on approximately 50 $^{o}$C, but further slowly increases. The behaviour of FC walls temperatures is very similar.

In modelling is taken into account, that after 10 minutes from beginning of the accident the operator takes actions directed on reduction of loss of the coolant through the break (following to the recommendations IAEA, in modelling is accepted non-intervention of the operator during the first 10 minutes). Is accepted that following the instruction [11], the operator closes valves on pressure and suction pipelines of MCP of affected loop. For preparation of valves for their closing is required not less than four minutes. Therefore, it was accepted in the modelling, that the specified valves are closed approximately in 14 minutes after beginning of the accident (i.e. at the moment of time t = 840 seconds). Thus the discharge of the coolant through the rupture from the MCP side stops. Only coolant flowing by the reverse flow from DS through GDH with fail to close check valve is discharged through the rupture.

As four ECCS pumps and three EFWP continuously supplying water to both loops of the MCC, after operator actions the pressure in DS and in the failed GDH begins slowly increase. The change of pressure is resulted in insignificant increase of the saturated steam flow rate through FC, connected to GDH with failed to close check valve (Figure 8). However is enough even of insignificant increase of flow rate for improvement of conditions of these FC cooling. The temperature of fuel cladding in the channels connected to failed GDH begin to decrease slowly after actions of the operator. Specified temperatures remain much below than safety criteria all investigated period of time. In modelling is accepted that at the beginning of the water cooldown, the operator, following the [10] disconnects one MCP from one side of MCC and one MCP from other side. It is assumed that at the same time occurred guillotine rupture of PH. During the first seconds after beginning of the accident because of the coolant flow rate decrease on hydrostatic bearings the electric motor of single operating MCP in the affected loop is disconnected. MCP in the intact loop of the MCC is disconnected by protection approximately in 680 seconds later. Due pressure difference decrease between PH and DS in the affected loop and increase of pressure in MCC compartments, the signal on ECCS activation is generated almost at once after rupture occurs. MFWP are in the switched off condition during reactor water cooldown state. Because of too high difference of pressure on fast acting valves on lines of water supplying from ECCS accumulators into GDH, these valves can not open. EFWP can not operate because of fast drop of pressure in the MCC also. Thus, on a signal about ECCS activation, only four ECCS pumps are supplying water into GDH of both loops of the MCC. After water temperature in MCC decreases down to 180 °C and pressure in DS up to 1 MPa, further reactor cooldown in water regime with removal of the heat in additional coolers of the purification and cooling system can bee choose. Reactor cooldown according to [10] should be done not exceeding speed of the water temperature change in the MCC more than 10 $^{o}$C/h. With the specified rate from the moment of AZ-1 operation prior to the beginning of water cooldown should pass not less than 10 hours. The reactor decay heat 10 hours after shutdown is 0.59 % of the initial power. If initial reactor power to accept equal 4200 MW, then in the PH guillotine rupture modelling, during water cooldown state the reactor power should be equal 24.8 $MW_{(th)}$.

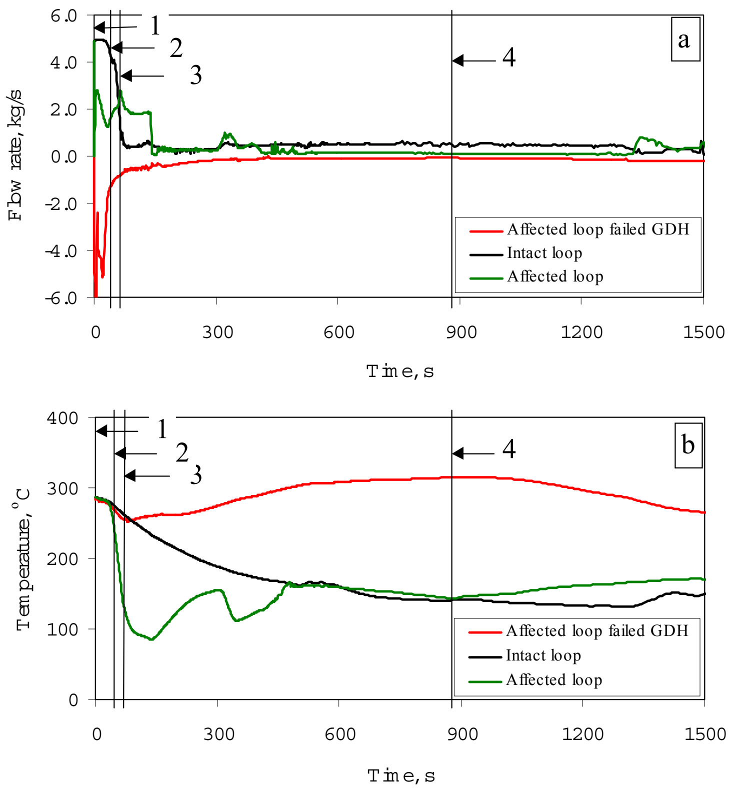


Figure 8 Pressure header rupture during steam reactor cooldown**:**
a - Coolant flow rate through fuel channels; b - Behaviour of the fuel cladding temperatures; 1 - Pressure header rupture, ECCS activation; 2 – Trip of MCP of intact loop; 3 – DS of affected loop get empty; 4 – Valves in the pressure and suction MCP pipelines are closed

The coolant supplied by operating MCP in the beginning of the accident cools the channels of intact loop of the MCC. After MCP switching-off, these FCs are cooled by ECCS water only. The channels of the affected loop of the MCC at once after pressure header rupture are cooled only by ECCS water (Figure 9). Fuel channels, connected to GDH with failed to close check valve is cooled by the reverse flow of the coolant from DS. Steam-water mixture flows through these channels in the beginning of the accident. However, after DS gets empty (after approximately 200 s from the beginning of the accident) these FCs are cooled by saturated steam only. While the channels are cooled by steam-water mixture, the temperatures of fuel cladding and FC walls drops, but after transition to cooling by saturated steam the temperatures begin slowly increase. After 500 seconds from the moment of the rupture, the temperatures of the fuel cladding are stabilised in the range of 200 $^{o}$C (Figure 9).

In modelling is taken into account, that after a bit more than 10 minutes after beginning of the accident, the operator takes actions directed on reduction of loss of the coolant through the break. It is assumed, that following the instruction [11], the operator closes valves on pressure and suction pipelines of MCP of affected loop. Thus the discharge of the coolant through the rupture from the MCP side stops. Only coolant flowing by the reverse flow from DS through GDH with fail to close check valve is discharged through the rupture. As four ECCS pumps continuously supplying water to both loops of the MCC, after approximately 1000 seconds after valves closing, DS of the affected loop start be filled by water. When the level of water in these DS exceeds a level of connection of steam-water communication pipelines, the water begins to flow into FC of the failed GDH. It results in sharp increase of the reverse coolant flow rate through GDH with failed to close check valve (Figure 9) and to sharp decreasing of fuel cladding temperature. The temperatures of fuel cladding and FC walls remain much below than safety criteria all investigated period of time.

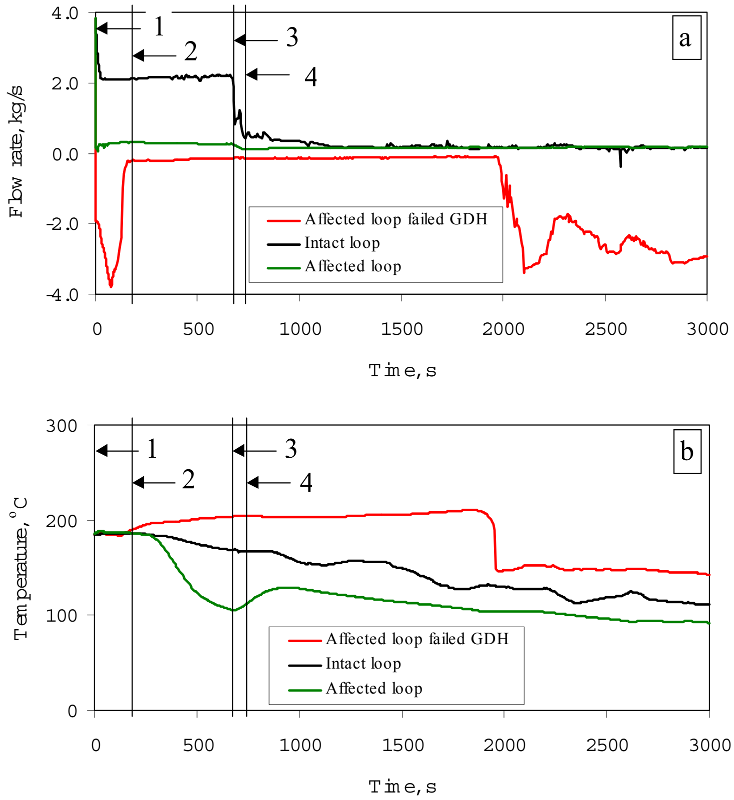


Figure 9 Pressure header rupture during water reactor cooldown**:**
a - Coolant flow rate through fuel channels; b - Behaviour of the fuel cladding temperatures; 1 - Pressure header rupture, ECCS activation; 2 – DS of affected loop get empty; 3 – Trip of MCP of intact loop; 4 – Valves in the pressure and suction MCP pipelines are closed

**4. CONCLUSIONS**

The INPP is unique among all RBMK type reactors in the scope and comprehensiveness of international studies that have been conducted to verify its design parameters and analyze its level of risk. Right from the start when Lithuania assumed control of the INPP (after the demise of the Soviet Union in 1991) the plant, its design and operational data has been completely open and accessible to western experts. Sweden provided initially effective assistance in the nuclear safety field, subsequently most states having significant nuclear expertise contributed.

International assistance took several forms, a very valuable mode of assistance utilized the knowledge of international experts in extensive international study programs whose purpose was: a) collection, systematization and verification of plant design data, b) analysis of the level of risk, c) recommendations leading to improvements in the level of safety, d) transfer of state of the art analytical methodology to Lithuanian specialists. The major large-scale international studies include:

- Barselina – A probabilistic risk analysis study conducted by Sweden, Lithuania and Russia.
- SAR – A very extensive international study funded by a grant from EBRD. Its purpose is to provide a comprehensive overview of plant status with special emphasis placed on its safety aspects. Specialists from the Ignalina NPP, Russia, Canada and Sweden contributed.
- RSR – An extensive review of the SAR by an independent group of international experts. Specialists from the U.S., United Kingdom, France, Germany, Italy, Russia and Lithuania contributed.
- SAR follow-up analyses

The noted studies provides a verified, state of the art base of knowledge which makes it possible to assess the present level of plant safety, compare this level with other reactor plants and plan improvements in plant hardware and operational procedures which enhance the level of safety. INPP is the only RBMK plant for which this information is available. Note that statements made regarding plant safety in this summary are based on the consensus reached in this area by the international expert community. A significant conclusion stated in the SAR is that none of the analyzed safety concerns require the immediate shutdown of the plant.

A strategy for destruction of local flow stagnation both in intermediate and long term has been developed. This includes an analysis of the loss-of-coolant accident conditions whose lead to the local flow stagnation in fuel channels and determination of the necessary actions such as ECCS injection management or main circulation circuit de-pressurization as well as the justification of the actuation of the new early ECCS initiation signal which compensates for stagnation flow. Thermal-hydraulic analysis was conducted using the state-of-the-art RELAP5 code. Results of analysis demonstrated that after implementation of the developed management strategy for destruction of local flow stagnation Ignalina nuclear power plant will be adequately protected not only in the case of the guillotine breaks of pipelines, but also following partial breaks which could result local flow degradation.

However, in spite that lot of safety improvements and analyses have been performed at the Ignalina NPP, much should be done in the nearest future.

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